Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/666
ISRAEL
February 1964
Submitted in:
ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

NEUTRON FLUX MEASUREMENTS IN A POOL TYPE LIGHT WATER-MODERATED RESEARCH REACTOR

B. Arad, G. Ben-David, E. Nardi and N. Pasternak

Israel Atomic Energy Commission

Measurements of thermal, epithermal and fast neutrons were carried out in the light water-moderated pool type reactor IRR-1. The purpose of these experiments was two-fold: evaluation of the thermal and epithermal neutron fluxes inside the core and the adjacent experimental facilities, and the study of the fast neutron spectral shape and its attenuation at various distances in the light water reflector.

The thermal and epithermal flux measurements were carried out using dysprosium foils as pure thermal detectors, and gold foils as mixed thermal and epithermal detectors. This technique was found to be simpler than other standard methods of flux measurement inside reactor cores, and yielded excellent agreement with these methods on the magnitude of the epithermal component.

The fast neutron spectra above 1 MeV were studied using fast neutron activation detectors. This method is the most convenient for fast neutron measurements inside reactors on account of its simplicity, adaptability and zero response to high γ fluxes. On the other hand, the wide energy response of each detector and the inaccuracy of the cross section data make the interpretation of the results difficult. Seven activation detectors were used, covering the region of about 1 to 15 MeV. The calculated spectrum was based on the simultaneous solution of the seven activation integrals.

The results indicate a hardening of the spectrum at increasing distances from the core. A dip in the spectrum between 3 and 4.5 MeV was observed at larger distances from the core and a smaller dip in the same region adjacent to the core. A slight hardening was also noted in the in-core spectrum as compared with a pure fission spectrum.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/667
ISRAEL
February 1964
Submitted in:
ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

REACTOR PHYSICS STUDIES OF NATURAL URANIUM-LIGHT WATER MODERATED LATTICES

W. Rothenstein**, G. Ben-David*, J. Helholtz**, Y. Segal** and D. Tabak**

* Israel Atomic Energy Commission

** Israel Institute of Technology, Haifa

The greater part of the investigation reported deals with the study of the influence of void channels surrounding the fuel on various reactor physics aspects of cylindrical uranium rods in light water.

The experimental part of the programme was carried out with the aid of an exponential assembly constructed adjacent to the thermal column of the Israel Research Reactor, which acted as a source of thermal neutrons. The hollow natural uranium fuel slugs, of 14 mm. internal and 28 mm. external diameter, were placed at the centre of aluminium tubes using aluminium spacers and leaving an air gap of 6 mm. width between the fuel and the moderator. The axial and radial flux distributions were measured experimentally by means of dysprosium foils and least squares fitted to the appropriate theoretical flux shapes to obtain the material bucklings. The lattices contain a considerable volume fraction of aluminium structural material and cladding. After applying appropriate corrections to allow for the effect of the aluminium, the buckling was found to reach a maximum of -2×10^{-4} cm^{-2} at a water to metal volume ratio of about 1.6.

The thermal flux distribution in the above lattices was calculated on the basis of monoenergetic transport theory. The boundary conditions previously applied in the spherical harmonics method to connect the moments of the angular flux at the two faces of the air gap were found to be non-unique and were replaced by a simpler set of conditions which made evaluation of higher P_L approximations possible. The calculations

of the ratio of the flux at the two faces of the gap were found to be in substantial agreement with experimental measurements. The predicted flux shape in the moderator was found to be flatter than the measured distribution. Discrepancies may be largely attributed to the neglect of thermalization effects in the calculations.

In the epithermal region, the Monte Carlo method was used as one of the principal tools for investigating the effects of void channels on the resonance capture. The Monte Carlo calculations covered energy ranges in which the main resonances of U^{238} are situated. The resonance escape probability in a lattice with air gaps was found to be somewhat larger than that in the corresponding lattice of equal volume ratio without gaps. This effect is due to the difference in the Dancoff correction in the two cases, which was determined separately and used in the semi-empirical formulae of the effective resonance integral for the lattices.

The Monte Carlo method was also used to check previous theoretical estimates of the contribution of the U^{238} p-wave resonances to the effective resonance integral. Theory shows that the contribution, which is independent of temperature, should approach the corresponding value at infinite dilution. While the Monte Carlo estimates confirm this trend, they also showed that the capture is affected to some extent by the close proximity of some p- and s-wave resonances, or p-wave resonances belonging to different values of the total angular momentum J of the compound nucleus.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/668
ISRAEL
February 1964
Submitted in: ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

MULTI-GROUP CONSTANTS FROM INTEGRAL DATA

Y. Humi, J.J. Wagschal and Y. Yeivin

The Hebrew University, Jerusalem, Israel

The G-group approximation in neutron transport calculations consists of replacing the correct, energy-dependent, Boltzmann equation, by the set of G-coupled equations

$$\vec{\Omega} \cdot \nabla \Psi_g + \alpha_g \Psi_g = \frac{1}{4\pi\gamma} \sum_{g'=1}^{G} \beta_{gg'} \varphi_{g'} \qquad g = 1,2,\ldots G$$

where Ψ_g and φ_g are the group angular and scalar fluxes, respectively. For simplicity, isotropic scattering was assumed. These equations involve $G(G+1)$ constant coefficients, the determination of which is usually a very delicate problem. (When anisotropic scattering is considered, the source terms are modified and involve more constants).

The common practice in determining these so-called group constants is to start from a set of basic cross sections which are then averaged with a proper weight function over the energy groups. However, there exists to date no satisfactory complete set of basic cross sections, and furthermore, one is not sure what the proper weighting function is.

Several unsuccessful attempts were made to derive sets of 'universal' group constants for many-group calculations. Certainly one cannot expect to derive by these 'conventional' methods reasonable constants for fewer-group calculations, applicable to a variety of systems. In principle it should be possible to derive the multi-group constants from integral data, rather than from the actual basic cross

sections. The purpose of this note is to illustrate and discuss this approach.

First should be decided what the purpose of a calculation is, and the quantities to be derived should be defined. Each such quantity forms a constraint on the manifold of possible sets of constants. If, for example, one is only interested in the eigenvalue (criticality), then obviously one still has many degrees of freedom in choosing one's constants.

The sets of constants for a certain system form a manifold in the proper hyperspace. Looking for a set of constants which will correctly describe several systems, one would naturally pick a set in the intersection of the relevant manifolds, if such an intersection actually exists. As more constraints, in the sense mentioned above, are imposed on our sets, the manifolds will shrink. As long as they still intersect, it will be possible to find a good set for all the systems.

This method is applied to several simple systems, and the derived constants are discussed and compared with the constants obtained by the conventional procedures.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/669
NORWAY
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

EXPERIMENTAL AND THEORETICAL STUDIES OF URANIUM OXIDE LATTICES MODERATED BY MIXTURES OF LIGHT AND HEAVY WATER

by

E. Andersen, D. Babala, J.O. Berg, K. Bryhn-Ingebrigtsen, V.O. Eriksen, H.R. Franzen, C.H. Millar, H. Moen, D. Perricos, J. Smit, R.J.J. Stamm'ler, K.V. Subba Rao, J.A. Thomassen, Z. Weiss

NORA Project, Institutt for Atomenergi,
Kjeller, Norway

Reactor physics measurements on UO_2 lattices moderated by mixtures of light and heavy water have been performed in the NORA zero power reactor at the Kjeller Research Establishment in Norway. The research programme is carried out as a joint project between the International Atomic Energy Agency and Institutt for Atomenergi, Norway.

The fuel used for these experiments has been UO_2 enriched to 3% U^{235} clad in stainless steel. The fuel was made available to the NORA Project by the United States Atomic Energy Commission.

A series of measurements on different H_2O/D_2O ratios and different moderator-to-fuel volume ratios has been completed. Macroscopic and microscopic neutron flux distributions have been measured using different types of foil materials resulting in sets of B_m^2 values and cell parameters for thermal and resonance neutrons.

The ratio of epi-cadmium to sub-cadmium capture in U^{238} (ρ^{28}), the ratio of epi-cadmium to sub-cadmium fission in U^{235} (δ^{25}), and initial conversion ratios have been measured using chemical separation techniques developed at Institutt for Atomenergi.

The ratio of Lu-foil activity to Cu-foil activity has been measured in different positions inside the fuel and in the moderator. The experimental results have been compared to calculated values of the same activation ratio and used as a spectral index for the lattices studied.

Supplementary to experiments on clean critical cores some measurements on two-region cores have been performed. Using the progressive substitution technique these experiments have, at the sacrifice of some accuracy, made it possible to extend the range of measured buckling values considerably.

Analysis of the experimental results using multigroup methods both for the fast and thermal neutron region will be presented.

The neutron energy spectrum in the fast region is calculated by a code BIGG which solves the Fourier transformed one-dimensional transport equation in the B-1 approximation using an extended Greuling-Goertzel treatment of the slowing down process, in 40 energy groups. Resonance absorption and fission are given a correct distribution in energy and are governed by input resonance integrals. BIGG is well adapted to handle slowing down in mixtures of light and heavy water.

The thermal part of the neutron spectrum is calculated by the K-7 THERMOS code which is a multigroup multiregion code solving the space and energy dependent integral transport equation in a heterogeneous reactor cell with cylindrical symmetry and with either reflective or white boundary conditions.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/670
PAKISTAN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

FAST-NEUTRON SPECTRA OF A POOL-TYPE REACTOR[1] AND A SHIELDED-DIODE SPECTROMETER

Dr. M.S. Bokhari
Pakistan Atomic Energy Commission

Measurements have been made of the spectra of fast neutrons within the core of a pool-type reactor, at the core-shield interface and at various positions in the water with a shielded spectrometer. Distances varied to 60 cm along the normal to the reactor face and to 40 cm at 51° and 40.5° from the normal. The data obtained along the normal to the reactor face show a hardening of the spectrum with increased distances in water and the presence of dips in the spectrum corresponding to resonances in the oxygen total cross section at 1.3, 3.7, and 4.1 to 4.3 MeV. These dips essentially disappear and the spectrum becomes softer at 51° and 41.5°. The data have been compared with theoretical calculations.

The shielded-diode spectrometer described herein has been found to be extremely capable of measuring fast-neutron reactor spectra in the midst of intense gamma-ray and fast-neutron backgrounds. The spectrometer consists of an independently supported Li^6F layer between two separated silicon surface-barrier diodes. Neutrons are detected by observing co-incidences between the alpha particles and the tritons produced in the $Li^6(n,\alpha)T$ reaction for which the Q value is 4.78 MeV. Substitution of a Li^7F layer for the Li^6F layer provides good accuracy in the measurement of the background from (n, charged-particle) reactions in silicon. The response of the spectrometer is linear over the neutron energy range from 0 to 14.7 MeV. The detection efficiency was measured to be ~ 3.5×10^{-3} for thermal neutrons and estimated to be 5.2×10^{-7} for 3-MeV neutrons.

1/ Work done at Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/671
PAKISTAN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

A REACTIVITY OSCILLATOR FOR SWIMMING-POOL REACTORS

M. Nuruddin Qazi
Pakistan Atomic Energy Commission

This paper describes in detail the development and testing of a very simple and reliable reactivity oscillator. Although it is particularly suited for swimming-pool reactors it can be used with other similar types of reactors either in the present form or with slight modifications. Its chief advantage is the ease and convenience in setting up and performing the actual experiment of amplitude transfer function measurement. The entire experiment of measuring the amplitude transfer function including the setting-up of the apparatus can be completed in about 4 1/2 hours which makes it particularly attractive as a teaching device. The device is capable of providing an infinitely adjustable frequency range from 0.01 c/s to over 100 c/s, which is achieved by using a variable speed DC motor coupled with a specially designed system of pulleys. The need for a long rotor shaft has been eliminated by the use of two selsyn servo motors which has resulted in considerable improvements. The reactivity oscillations have been achieved by oscillating the amount of cadmium from a minimum to a maximum. This has been accomplished by rotating an aluminum disc carrying symmetrically placed cadmium sheets over another stationary aluminum disc exactly similar to the former. The resulting fluctuations in the neutron density of the core are detected by an ionization chamber and displayed on an oscilloscope. A polaroid camera is used to record these fluctuations from the oscilloscope screen. This apparatus was used to measure the amplitude transfer function of the Pennsylvania State University Reactor.[1] The experimentally obtained amplitude transfer function plots were compared with those computed theoretically on IBM-7074 for various values of ℓ^* - the prompt neutron lifetime.

1/ This work was done for the Masters degree at The Pennsylvania State University, University Park, Pa., U.S.A.

In computing the theoretical results, a value of 0.008 was used for the parameter $\bar{\beta}$ - the total effective fraction of the delayed neutrons - based on the experimental determination in BSR-1 at the Oak Ridge National Laboratory. The prompt neutron lifetime in the Pennsylvania State University Reactor was thus found to be 65 micro-seconds.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/672
POLAND
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

IMPORTANCE OF THE CLASSICAL SUPERHEAT IN THE NUCLEAR POWER STATIONS IN THE NEXT DECADES

St.Andrzejewski
Technical University of Warsaw

It could be stated that in the next decades the nuclear power stations will be constructed simultaneously with the classical power stations. The studies and calculations have demonstrated that, until the technology of high temperature reactors will be completely solved, it will be more economical to construct combined power stations in which heat from the reactor will serve only to evaporate water, and fossil fuel will be used to superheat the steam. In this case reactors could be constructed for medium temperatures. The studies of the availability of different types of the reactors have proved, that only the medium temperature reactors are in the present time really available. It is well known that pressurised water reactors /for instance Indian Point/ and boiling water reactors can work with the simple superheat. The paper shows that the boiling water graphite reactors /Bielojarsk type/ and the reactors cooled by organic liquid or by gas can also economically work in the scheme of superheat and reheat. The conception of the graphite moderated, carbon dioxide cooled Magnox type reactor in the scheme of supercritical pressure and double reheat was analyzed. In this scheme the reactor heat exchanger produces supercritical steam for the main turbine. There exists a second separate low pressure circuit for the auxiliary propulsion /carbon dioxide blowers and feed pumps/. The influence of the superheat on the cost of the turbosets is analysed. The preliminary design and calculations have proved, that incremental power done by superheating is cheaper than in the conventional power stations and the efficiency of the fuel burning is better. The conclusion of the paper is the following: If the nuclear and classical power stations will be constructed simultaneously in any country, the best economical solution in the next decade might be the construction of combined nuclear - conventional power stations i.e. nuclear steam generator with the conventional superheat.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/673
ROMANIA
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

THE TEMPERATURE DEPENDENCE OF THE NEUTRON INELASTIC SCATTERING ON HEAVY WATER

N. Mateescu, H. Teutsch, V. Nahorniak, A. Diaconescu and P. Timis
Institute of Atomic Physics, Bucharest, Romania

The scattering of cold neutrons on heavy water was studied, using the filter technique for producing the cold neutron beam and the time of flight method for the analysis of scattered neutron spectra.

For the achievement of an optimal ratio of cold neutrons to thermal and epithermal ones we chose a filtration assembly formed by a beryllium policrystalline filter and a lead monocrystal.

The over-all time of flight resolution of the device of 3.5% has been obtained using a curved slit chopper in optimal transmission regime for 4 Å neutrons.

The measurements were extended over an angular range of 30° to 90°, the sample temperature varying between -100°C and +80°C.

The obtained curves have been corrected for background, chopper transmission, scattering and absorption in air and counter efficacity.

The diffusive properties of heavy water dependent on temperature have been obtained by detailed analysis of the quasielastic scattering peaks.

The results have been compared with theoretical models viz. with Vineyard's model (continuous diffusion) and Singwi-Sjölander model (jump diffusion) respectively and also with the experimental results obtained in other laboratories.

A quantum level superimposed on the quasielastic peak was also detected; it is probably due to a fine structure having the same order of magnitude as that obtained by Hughes et al. in the case of light water.

In the inelastic scattering region a succession of peaks have been observed which have been ascribed to the heavy water molecule rotation vibration levels.

As the target temperature decreases down to -100°C there occurs a significant modification of the inelastic scattering spectra.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/674
ROMANIA
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

OPTIMALIZATION OF THE INITIAL CONVERSION RATIO OF A NUCLEAR REACTOR CONSIDERED AS A GAMES THEORY PROBLEM

I.I. Purica
Atomic Physics Institute, Bucharest, Romania

It is shown that the problem of obtaining a maximum initial conversion ratio (by non-homogeneous spatial distribution of the fissile material), together with minimum fuel consumption, has the structure of a two-matrix game with a non-zero sum.

The volume of the reactor is regarded as being divided into q separate, indeterminate volumes, each containing $U_i = \mu_i U$ fertile nuclei and $P_i = \xi_i P_c$ fissile nuclei, where U is the total number of fertile nuclei and P_c is the number of fissile nuclei necessary for critical mass.

These conditions of maximum density and criticality make it possible to determine the geometrical structure and the value of P_c for a given U and a given distribution (μ_i, ξ_i).

The number of fissile nuclei produced in a reactor in unit time is

$$K_1 = \sigma_u^{cp} \cdot \sum_{i=1}^{q} \cdot \mu_i \cdot \bar{\phi}_i$$

and the number of fissile nuclei disappearing is

$$K_2 = \sigma_p^a \cdot p \cdot \sum_{i=1}^{q} \cdot \xi_i \cdot \bar{\phi}_i$$

where σ_u^{cp} is the cross-section of the process leading to the appearance of a fissile nucleus, σ_p^a the cross-section of the process leading to the disappearance of a fissile nucleus, $\bar{\phi}_i$ the mean neutron flux in the region i, and $p = P_c/U$.

The gain K_1 and the loss K_2 of fuel are functions of the distribution (μ_i, ξ_i) adopted and of U.

The ratio K_1/K_2 is the breeder ratio of a breeder reactor.

Let us assume two players, one possessing U fertile nuclei and adopting the distribution μ_i^k, the other adopting the distribution ξ_i^e of P_c fissile nuclei.

The gain of the first is given by the terms of the matrix $K_1 (\mu_i^k, \xi_i^e)$ and the loss of the second is given by those of the matrix $K_2 (\mu_i^k, \xi_i^e)$. The first is therefore interested in choosing the μ_i^k distribution which makes K_1 a maximum for any μ_i^e distribution chosen by the second. The second will choose the ξ_i^e distribution making K_2 a minimum for any μ_i^k distribution chosen by the first.

The problem thus corresponds to the structure of a two-matrix game with a non-zero sum $\Gamma (K_1, K_2)$.

For fast neutron breeder and two-region thermal reactors, an analysis is made of the information that can be obtained by studying the strategy of the game.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A.675
ROMANIA
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

DETERMINATION OF THE DEFORMATION OF THE THERMAL NEUTRON SPECTRUM IN A MULTIPLYING LATTICE USING THE FREQUENCY CHARACTERISTIC METHOD*

I. Purica, N. Seferian and E. Răcătăianu
Atomic Physics Institute, Bucharest, Romania

The paper analyses the possibilities offered by the frequency characteristic method for determining the deformation of the thermal neutron spectrum in a multiplying lattice. Use is made of the P_1L_2 approximation.

The theoretical results are compared with experiments carried out on an enriched uranium-graphite subcritical assembly. Separation of the effect of absorption in the uranium on deformation of the neutron spectrum is achieved by substituting suitable absorbing and diffusing substances for uranium.

* Work carried out under International Atomic Energy Agency Research Contract 229/RB.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/676
ROMANIA
February 1964
Submitted in:
FRENCH and RUSSIAN

Confidential until official release during Conference

Abstract

FUEL BURN-UP IN THE WWR-C-BUCHAREST REACTOR

I. Udrea, L. Mănțescu, P. Popa and D. Cepraga
Atomic Physics Institute, Bucharest, Romania

Fuel burn-up is studied on groups of 16 rods, and is defined by the function

$$FA_i = - \frac{f}{k} \frac{\Sigma ai}{\Sigma' ac}$$

where Σ ai is the absorption cross-section in the fission products and Σ' ac is the total absorption cross-section in the burned-up uranium. Variation of the factor η with burn-up is neglected and justification for this is given.

The quantity of uranium-235 burned up, the absorption cross-section of the fission products and variation of the factor η f with burn-up are deduced from these functions as measured on the WWR-C reactor using the danger coefficient method.

These values are compared with those deduced from the poisoning characteristic.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/677
ROMANIA
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

A SYSTEM OF AUTOMATION FOR THE WWR-C-BUCHAREST REACTOR

I. Iftode, C. Dobrescue and C. Tapu
Atomic Physics Institute, Bucharest, Romania

The paper describes a system of automation of the principal operational functions of research reactors (in particular the WWR-C Reactor at Bucharest). The chief functions considered are: start-up from zero power; stabilization and operation at a given power level; transition from one power level to another; operation in accordance with a particular power programme; protection of the reactor when a given power level is exceeded.

For start-up, the variation in the signal proportional to the reactor power (8-10 decades) is compressed by means of operational amplifiers whose gain is automatically adjusted (feedback resistors) as a function of the input signal (ionization chamber current). The output voltage of the amplifiers is automatically kept between the limits U_{min}-U_{max} so that there is no need for logarithmic amplifiers over a wide range of power variations.

For start-up and for large power variations the useful control signal (error) is:

$$e_1 = k_1\left(\frac{1}{T} - \frac{1}{T_o}\right)$$

where T is the reactor period.

Since the ratio $\frac{Umax}{Umin} = 5$ is less than one decade, a simple diode function-generator can be used as logarithmic amplifier, the logarithmic function being approximated by three segments.

For stabilization and operation at a given power level $\left[P_o - dP_o < P < P_o + dP_o\right]$ or when the power change is slow, the useful control signal is:

$$e_2 = k_2(P-P_o) + k_3\frac{dP}{dt}$$

The signal required is also chosen automatically as a function of the value of $dP = P-P_o$.

During start-up or during large changes in power level the reactor is protected by the period, and during operation at a given power level protection is achieved by exceeding the level.

Compared with other systems, the one described has the advantage of employing a small number of relatively simple components, thus simplifying checking and maintenance work. The use of operational amplifiers makes it possible to add other computing elements to the system.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/678
SPAIN
February 1964
Original: SPANISH

Confidential until official release during Conference

Abstract

SOME APPLICATIONS OF THE PULSED NEUTRON SOURCE TECHNIQUE

F. Verdaguer, F. Cordero, B.G. Castañer, E.R. Mayquez, M.A. Vigón
Junta de Energía Nuclear. Madrid

In this paper we present some applications of the pulsed neutron source technique, the results obtained and the measurements in progress.

A 600 Kv Cockroft-Walton accelerator has been used as a neutron source. The neutrons bursts are obtained pulsing a Penning type ion source.

The first measurements have been performed with a 9 channels time analyser, but, at present, it is in use a new 256 channels transistorized instrument.

First, the diffusion parameters of light water have been measured. The results agree with the published ones.

The connection between the cooling effect coefficient, c, and the mean square energy transfer, M_2, suggested the possibility to perform a measurement of M_2 through the perturbations which are produced on the capture and diffusion constant of non-1/v capturing media. This media have been obtained dissolving differents amounts of cadmium in light water. Those perturbations depends on the cadmium concentration, on the temperature dependence of the effective cross-section of cadmium and on M_2. From this measurements a value, $M_2 = 1,65 \pm 0,49\ cm^{-1}$, is deduced.

Previous results from measurements of the diffusion para-

Translation provided by courtesy of the submitting Government

meters of Santowax-R in function of the temperature are, also, presented. To calibrate the experimental set up used, measurements have been performed in which Gilotherm substitutes Santowax-R.

With regard to non-multiplying media, measurements are in progress to study the effect produced by a reflector on the mean life of neutrons in moderators.

In order to apply the pulsed neutron source technique to multiplying media, a 150 Kv pulsed accelerator has been built. The firsts measurements will be performed on a heavy water moderated subcritical assembly and with two different kinds of fuel elements. Uranium oxide and uranium carbide organic cooled fuel elements. The measurements will be performed at ambient temperature. This subcritical assembly is described in another paper to be presented to this Conference, relative to measurements with stationnary neutron sources.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/679
SWEDEN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

EXPERIENCE FROM THE COMMISSIONING, THE CRITICALITY EXPERIMENTS AND THE POWER OPERATION OF THE ÅGESTA NUCLEAR POWER PLANT

N. Rydell[1/], P. Blomberg[1/] and E. Ericsson[2/]

The paper presents programmes of and experiments from the three major phases in the commissioning of the Ågesta Nuclear Power Plant.

Light water commissioning

The light water commissioning was carried out at room temperature from 1 December 1962 to 31 January 1963, and at higher temperatures from 1 February to 31 May.

The water rinsing brought out 2 grammes of dry material from the primary circuits.

The primary circuits and components performed well except in the following respects:

Valve operation by motors or manually via mechanical links was unsatisfactory;

One of the main pumps failed because of a break of a stator winding;

The electric heaters in the pressurizer failed by overheating.

Many adjustments were required before the light water circuits operated satisfactorily. Inleakage of air in a cooling circuit working at under-pressure was particularly difficult to overcome.

Criticality experiments

The criticality experiments were carried out at room temperature from 1 July to 31 October and at operating temperature from 1 November to 31 January 1964.

1/ AB Atomenergi, Stockholm.

2/ State Power Board, Stockholm.

The primary circuits were loaded with 76 123 kg heavy water of 99.825±0.002% D_2O-content before and 99.825±0.005% after the loading. Thus less than 5 kg light water remained from the commissioning.

All relevant control and safety equipment was connected during the criticality experiments. The fuel elements were successively loaded into the heavy water fuelled reactor tank. Neutron detectors were positioned in the core together with a compact neutron pulse source, which was used for determination of subcritical conditions from the neutron pulse decay rate. Spontaneous fissions in the fuel gave sufficient source neutrons.

The first criticality was achieved on 17 July with 23 (of totally 140) fuel elements loaded. Excess reactivities, control worths, reactivity coefficients of water level and neutron flux distributions were measured at three core sizes at room temperature. The effect of temperature on the reactor physics properties and the power distribution at different control rod configurations were determined with the full core.

The conservatively projected number of control rods could be considerably reduced as a result of these measurements, which also gave a valuable test of the new theoretical treatment of control rods in large reactor systems.

Primary circuit equipment and the instrumentation performed well, whereas the power supply equipment gave repeated though minor problems. Fourteen unscheduled scrams occurred from 15 July to 15 January, all due to equipment failure or personnel error.

The heavy water degradation, if any, lay within measuring uncertainty ±0.005%.

The heavy water losses were less than measuring uncertainty ±80 kg. Observed losses from loading operations and from a leaking valve amounted to less than 20 kg.

Power operation

The first approach to power will be made in four steps, resp. 1%, 20%, 50% and 100% of full power, 65 MW.

Experience of particular interest will be presented within the following fields:

Reactor Physics
Control and Instrumentation
Thermal Plant Engineering
Chemistry
Health Physics
Waste Disposal.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/680
SWEDEN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Investigations of the Moderating Properties of Water

G. Grosshög [1], K.-E. Larsson [2], E. Möller [3], S. Purohit [3], N. G. Sjöstrand [1]

A review is given of the work performed and going on in Sweden to investigate the moderating properties of water by various experimental and theoretical methods.

Using slow neutron time-of-flight spectrometers the nature of atomic motions in light and heavy water has been investigated, in the case of light water in the temperature range from -3 to +95°C and in the case of heavy water from -10° to +300°C. Generalized frequency distributions have been derived from the experimentally determined inelasticly scattered neutron spectra and found to exist even at +300°C. This fact as well as the width of the quasi-elastic peak indicates a clear solid-like behaviour of the water molecules. It is shown that the continuous diffusion model does not explain the width behaviour and that the gas model does not describe the translational motions of the water molecules. It is found that the quasi-elastic peak is to a high degree determined by the motion of the protons in the hydrogen bonds.

The time behaviour of neutrons during slowing down and thermalization in water has been studied by the use of a pulsed van de Graaff. Information on the change with time of the neutron spectrum was obtained from its reaction with spectrum indicators, the reaction rate being observed by the detection of capture gamma rays. Measurements were made with cadmium, gadolinium and samarium as indicators. A slowing-down time to 0.2 ev of 2.7 ± 0.4 μsec was obtained. From 9 μsec after the injection the results are well described by

1) Chalmers University of Technology, Göteborg
2) Royal Institute of Technology, Stockholm
3) AB Atomenergi, Studsvik

the assumption of the flux as a Maxwell distribution cooling down to the moderator temperature with a thermalization time constant of 4.1 ± 0.4 μsec.

The slowing-down time of Ra-Be neutrons in water to the cut-off energies of cadmium and gadolinium filters has been measured by a stationary method. The slowing-down time is derived from the ratio of the integrated neutron densities measured in a large medium using a 1/v-detector without and with the filter in question. The slowing-down times obtained are 1.60 ± 0.07 and 2.47 ± 0.11 μsec to the cut-off energies 0.5 and 0.3 ev respectively.

Various theoretical models for neutron slowing down in water have been compared with the integral experiments. A numerical solution of the Boltzmann equation for the proton gas model shows that the validity of this model is limited to energies above 0.4 ev. The Nelkin model is better, but the agreement with experiments is still not satisfactory. The possibilities of improvements based on the detailed differential measurement are discussed.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/681
SWEDEN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

A Method for Calculating Collision Probabilities in General Cylindrical Geometry and Applications to Flux Distributions and Dancoff Factors

I. Carlvik [1)]

The collision probability method for calculating neutron flux distributions is at present acquiring increasing attention particularly for its application to comparatively small strongly heterogeneous systems, for example reactor lattice cells. This method can be applied to systems with rather complicated geometry. Due to the fact that it ignores the angular flux distribution and concentrates on the interesting quantity, the scalar flux, it can in many cases give a considerable saving in computer time compared to various other methods for numerically solving the transport equation. An obvious drawback is that this method usually assumes isotropic scattering and regionwise constant flux.

The main feature of the method is the calculation of the collision probabilities. Fundamental equations for cylindrical geometry are presented in this paper, and are applied to cells with annular geometry and to cells containing a central rod cluster. The necessary integrations are performed in a way different from the one commonly used, in order to shorten the numerical work. Particularly for the annular geometry, it is possible to arrange the integration in such a way that an efficient Gauss integration can be performed. The Gauss constants are chosen with regard to the known irregularities of the integrand, with different possible sets of constants being investigated.

Moments of the first flight distance can be calculated in much the same way as the collision probabilities for neutrons originating in each region. This makes it possible to calculate the axial and radial diffusion areas for a given source distribution when the flux distribution has been solved. This theory is also presented in the paper.

1) AB Atomenergi, Stockholm

This work has lead to several computer programs, and results from calculations are reported in the paper. Since the problem under investigation is the transport of neutrons, only one energy group problems have been investigated so far. However, there is no objection to the use of these methods in many-group calculations.

Only a white boundary condition has been used throughout the work in this paper at the cell boundary. The collision probability matrix is calculated for a black cell boundary, and the white boundary matrix is obtained by a simple adjustment.

A closely related problem would be the calculation of Dancoff factors. In the case of annular geometry, Dancoff factors can be obtained directly from collision probabilities. Dancoff factors can be obtained in the same way for cluster geometry, but it would be more suitable to use a special computer program. In this paper a theory is presented for the calculation of Dancoff factors in cluster geometry, and tables of results are given for the ordinary 7-, 19-, and 37-rod cluster.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/682
SWEDEN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Reactor Physics Studies and Comparison between Reactor Physics Data from Calculations and Mock-up Studies and from Measurements in the Ågesta Nuclear Power Plant

G. Apelqvist [1], P.E. Blomberg [2], O. Nylund [3], R. Persson [2]

At the time when the Ågesta reactor was designed, the only direct experimental information of relevant reactor physics data available was obtained from exponential measurements. The calculations were based on diffusion theory and four-factor formalism.

Complementary data has since been produced by new experimental facilities and calculational methods.

The 19-rod clusters were investigated in exponential and critical assemblies both in Sweden and at the Savannah River Laboratory in USA. The buckling was measured as a function of pitch, internal spacing in the cluster, and temperature. Void effects were also studied and intercell flux measurements made. The results of these measurements did not cause any appreciable change of the calculated neutron economy and hence of the design.

In the Ågesta reactor itself critical water levels were determined with three core sizes, with and without control rods. Bucklings and reflector savings were obtained from copper wire activations, and the reactivity loss due to temperature increase was measured up to 215°C with control rods in the core.

Microscopic properties in the lattice cell were investigated by means of activation techniques. The studies included determinations of neutron

1) State Power Board, Stockholm
2) AB Atomenergi, Stockholm and Studsvik
3) ASEA, Västerås

spectrum parameters, the initial conversion ratio, disadvantage factors and the flux peaking in the fuel pin connections.

The control rod worth and their influence on the power distribution have been calculated with homogeneous and heterogeneous methods. The extensive series of control rod measurements, both in critical and exponential assemblies and in the power reactor itself, served as a good base for correlation with the theory. The reactivity worths were obtained from measurements of bucklings and reactivity water-level coefficients, and from neutron pulse decay curves. The perturbation of the power distribution due to control rods was studied by activation techniques and movable detectors.

Dynamic characteristics are to be studied as a function of power level and burn-up by oscillatory perturbations and by noise analysis. Time constants and reactivity coefficients will be obtained by fitting an analogue model to the measured transfer functions.

A thorough preparation has been made for investigating the burn-up effects in the fuel elements, of which a representative set of pins have been inter-calibrated with the pile-oscillation technique. The changes of the excess reactivity and the neutron spectrum parameters in the fuel, as a function of the neutron dose, are to be included in the studies of the burn-up effect.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/683
SWEDEN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Theory and Applications of Heterogeneous Methods for Heavy Water Lattice Calculations

A. Jonsson [1], P.-E. Ahlström [2], P. Lindgren [3], G. Näslund [1], T. Skardhamar [1]

Commonly used methods for overall reactor calculations of reactivity and flux distributions assume that the reactor media can be considered as homogeneous. It has been shown by Galanin that this is a valid assumption for a heterogeneous reactor if the lattice is uniform and the number of fuel elements sufficiently large. For defective or composite lattices containing for example vacant positions, fuel elements of different types or control rods, or if the number of elements is less than 20-30, homogeneous theory may apply less well. To handle these cases a series of heterogeneous computer programmes have been developed at AB Atomenergi. Comparisons with experiments have given some confidence in these methods for treating uniform as well as non-uniform heavy water lattices. The purpose of this paper is to summarize the experience and present status of the work, and to describe how the heterogeneous method has been adapted to heavy water power reactor calculations.

The paper is divided into four parts. Part one outlines the basic theory for a finite cylindrical reactor in the three-dimensional, two-group diffusion theory case. The basis of the heterogeneous method is an analytical solution of the diffusion equation assumed to hold in the moderator separating the different elements of the core. This general solution is subject to boundary conditions on the moderator surface which is taken to consist of an outer boundary in the form of a circular cylinder of finite height, and a number of disconnected inner

1) AB Atomenergi, Stockholm
2) Swedish State Power Board, Stockholm
3) ASEA, Västerås

surfaces, the fuel-moderator interfaces. At the outer boundary the zero flux condition is imposed, and at the inner boundaries extrapolation length type conditions are used.

In part two, different numerical methods for the solution of the basic equations are discussed, mainly in the three-dimensional case. An outline of the existing computer program is given. The numerical method is also briefly compared to the conventional difference technique with respect to calculating time and storage requirements.

In part three, the application of heterogeneous methods for predicting reactivity effects and flux distributions in heavy water lattices is discussed and compared with experimental measurements in the R0 critical facility and the Ågesta power reactor. An outline is given of the accuracy with which present methods can be applied for calculations on both uniform and non-uniform lattices, in particular for determining the reactivity effects and flux distorsions of single and groups of control rods.

Part four describes some applications of heterogeneous methods to design calculations for the Marviken power reactor, including studies of different fuel cycles, different ways of initial loading, and of control rod effects.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/684
SWEDEN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

A New Shield Design Method and Some Attenuation Studies in Laminated Biological Shields

E. Aalto [1], L. Hjärne [1], M. Leimdörfer [2]

A computer method has been developed to calculate the penetration of neutrons and gamma-rays through bulk shields. The attenuation and slowing-down of neutrons is treated by a combination of first-flight removal and multigroup-diffusion theories. The energy-dependent flux of such fission neutrons which are uncollided or have suffered only small-angle collisions is calculated throughout the shield; this flux provides source terms in a series of multi-group diffusion equations. Slowing-down between the diffusion groups is treated by a discrete-collision formalism in which the correct average energy loss of a neutron in a specific energy group is taken into account. Inelastic scattering and moderation by light nuclei come in naturally as slowing-down between all groups is permitted.

The transport of gamma radiation born in the reactor core and in the shield is calculated with a Monte Carlo scheme employing various variance reducing techniques, e. g. the exponential transformation whereby the outward directed motion of photons is enhanced and the slope of the source distribution is modified to be more favourable for Monte Carlo treatment.

The whole method is coded for the Ferranti Mercury computer. It includes an initial part which provides all the required neutron cross-section data. The gamma transport section has been translated to FORTRAN and exists in a version for the IBM 7090 computer.

The main advantage of the method presented is that it requires only a modest amount of computer time and capacity, compared to other systems giving the

1) AB Atomenergi, Stockholm
2) Research Institute of the National Defense, Stockholm

same quality and accuracy of results.

An extensive series of attenuation measurements in bulk shields has been performed at the Studsvik research centre. The radiation source has been the R2-0 reactor, furnished with special facilities for shielding studies. Measurements have been made up to $\sim$ 2 m penetration depths in a number of configurations of water - magnetite concrete - lead, in two configurations of water - ordinary concrete and, up to 0.5 - 1 m penetration depths, in configurations of layered iron and heavy water. Both neutron and gamma fluxes have been measured throughout the shields.

In order to study the accuracy and reliability of the method described, calculations have been performed in the same configurations, using the best available input values. For comparison, some calculations have been performed also with 1) an earlier 18-group removal method combined with a build-up factor treatment of gamma transport; 2) with a method calculating neutron fluxes by numerical integration of the Boltzmann equation; 3) with a neutron Monte Carlo code.

All of the methods mentioned above give a rather good agreement with the measured values, normally within a factor 2 - 3 after $\sim$ 2 m penetration.

As for the safety factors to be used when designing massive biological shields, we can draw the conclusion that the accuracy of the best available methods is so good that the uncertainty originating from other factors (e.g. radiation sources, concrete density) gives errors of the same order of magnitude as the calculation methods do.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/685
SWEDEN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Spatial Kinetics of Thermal Reactors

O. Norinder[1)]

In Sweden the theory of reactor spatial kinetics is studied and developments attempted to make the theory more clear and simplify its interpretation. Studies that elucidate the kinetic properties of the Marviken reactor are given particular attention. To be significant for this reactor the theory must be valid for macroscopically heterogeneous cores. The actual description of the reactor flux and other variables is mostly founded on the one-group time-dependent neutron diffusion equation. By expansion in the flux-mode system one obtains a graphical description of the time behaviour. The significant part of the transient is represented by the fundamental mode and the three higher modes. From the velocity characteristics of the transients one might group them as prompt jump -, fuel heat transport and delayed neutron -, external heat circuit -, xenon -, and depletion transients.

In kinetic investigations of reactors one generally uses several stability concepts. These must be defined. For a reactor the complete system always must be stable. Spatial stability conveniently is defined as stability without a control system or with a control system actuated by one real variable. Xenon spatial stability may be investigated using flux eigenfunctions. Computed tables of such functions are described.

A new theory of void distribution spatial stability is given. The theory makes possible numerical estimates of the stability conditions. In all cases calculated including unstable situations the flux part of the least stable eigenvector had only a small content of non-fundamental mode flux. In all unstable cases computed only this eigenvector had a positive real part eigenvalue.

1) AB Atomenergi, Stockholm

This fact indicates that in many investigations a space - time separated treatment of the flux is possible and that spatial instability in combination with fundamental mode stability is unlikely.

For calculation of transients it is mostly necessary to separate the space and time dependence. It is mostly possible to retain the significant characteristics of transients important to the reactor design when space - time - separated description is used. A short description is given of a code for calculating spatial transients. The possibilities of estimating spatial transient behaviour from static flux calculations are investigated. The demands for calculations or reactor spatial transients are discussed.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/686
SWITZERLAND
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

THE EVOLUTION OF FISSION RATE DURING AND AFTER SHUTDOWN OF A NUCLEAR REACTOR

by P. Schmid and W. Zünti
Swiss Federal Institute for Reactor Research,
Würenlingen / Switzerland

The evolution of the fission rate during and shortly after fast shutdown of a nuclear reactor may contribute essentially to the total heat released to the fuel and is, therefore, subject to estimates both by the design engineers and by safeguard officers. Furthermore the shape of the function representing neutron flux at any location versus the time elapsed after sudden insertion of control absorbers contains essential and reliable information about reactivity in the particular shutdown state. The latter fact has been explored, e.g., by recently developed reactivity measurement techniques called "modified rod drop method".

The present paper summarizes useful knowledge about shutdown phenomena acquired at this institute while carrying out computations and experimental programmes of the types mentioned above. Upper and lower bounds are found for the fission rate during the finite travel time of the shut-off device. From this an equivalent reactivity step is defined such that the asymptotic evolution of fission rate is matched. For the fission rate after the shut-off device has reached its final position, an expansion into a series of characteristic functions of time is derived, the respective coefficients being given by the powers of the reciprocal reactivity in dollar units. This expansion converges well for final reactivities below -1 $ and proves very useful, not only for quick estimates of the fission rate, but also for optimum analysis of "modified rod drop experiments". An approximation

Translation provided by courtesy of the submitting Government

for the total fission energy E released after the initiation of a fast shutdown is presented for sub prompt critical excursions with initial periods T_o not shorter than the (properly defined) travel time θ of the shut-off device, the latter being driven in with constant acceleration.

The contribution E_I which would be observed with instantaneous, stepwise reactivity reduction from ρ_o to $\rho_o - |\Delta\rho|$ is given exactly by

$$E_I = \frac{r(0)}{|\Delta\rho| - \rho_o}\,\rho_o T_o ,$$

where $r(0)$ is the instantaneous fission rate at the releasing time of the shut-off device. The nontrivial contribution to E due to the inertia of the shut-off device, is defined by $E_\theta = E - E_I$. Replacing the actual regulating curve (reactivity versus travel distance) by a 2^{nd} order parabola well fitted near the initial position leads to the following bounds for E_θ:

$$\frac{r(0)}{|\Delta\rho| - \rho_o} \cdot \frac{\theta}{1 - \frac{\tau}{\rho_o T_o}}\, M_o\left(\frac{|\Delta\rho|}{1 - \rho_o}\right) \bar{D}\left(\frac{1}{T_o}\right) \leqslant E_\theta \leqslant \frac{r(0)}{|\Delta\rho| - \rho_o} \cdot \frac{\theta}{1 - \frac{\tau}{\rho_o T_o}}\, M_o\left(\frac{|\Delta\rho|}{1 - \rho_o}\right),$$

where τ is the reduced generation time, $\bar{D}(s)$ is the Laplace transform of the normalized delay distribution function and $M_o(x)$ is a non-elementary function specified and graphically represented in this paper.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/687
SWITZERLAND
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

ON THE STABILITY OF STEAM-COOLED FAST REACTORS WITH CERAMIC FUEL

by P. Schmid
Swiss Federal Institute for Reactor Research,
Würenlingen / Switzerland

By virtue of their short generation times fast reactors may be considered as particularly safe, provided that their reactivity feedback is governed by a prompt negative contribution from effects inherent to the system. In large ceramic fueled reactors this is to be ensured by the Doppler coefficient of reactivity. It is generally known that stability may be jeopardized by sufficiently large delayed contributions of either sign to the total reactivity feedback. Nevertheless one has to cope with the fact that Uranium-Plutonium oxide fueled fast breeder reactors with core volumes larger than a few hundred liters exhibit sizable positive void coefficients if they are cooled by steam. The present paper derives stability criteria which allow to determine the maximum permissible ratio of magnitudes of the positive steam void coefficient and the negative Doppler coefficient of reactivity. Two models are considered, according to the types of instability to be prevented.

The first one assumes Newtons law of cooling with coupled heat transfer, but neglects the influence of those parts of the closed steam loop which are outside the reactor core. From a theorem by Popov necessary and sufficient conditions for asymptotic stability in the large are obtained. These conditions state that

a) the equilibrium net power coefficient must be negative,

b) a power pulse must result in an initial drop in reactivity,

Translation provided by courtesy of the submitting Government

c) redistribution of heat following the power pulse must not cause any further decrease of reactivity.

According to the inherent limitations of the first model the satisfaction of these conditions is considered to ensure against selfsustained oscillations with periods comparable to or shorter than the coolant transit time through the core.

The second model of reactivity feedback is more realistic if possible instabilities at intermediate and lower frequencies are to be considered. Heat transfer from the fuel element to dry steam as well as the performance of out of pile parts of the coolant loop are described in a rather general and flexible manner by a series of characteristic parameters, valid at least in the neighbourhood of the considered equilibrium state. Applying the First Stability Theorem by H.B. Smets, a necessary condition for the asymptotic stability in the large is derived for such a system with linear reactivity feedback. Using some approximations well justified in the light of published design characteristics of a 300 MW_e steam cooled fast breeder reactor power plant, a simplified version of the former criterion is found: the ratio of the Doppler coefficient to the steam void coefficient is greater or equal to an elementary expression containing only two essential characteristics, which are an average relaxation time for heat removal from the fuel and a suitable parameter which characterizes the efficiency of heat removal from the primary steam loop.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/688
SWITZERLAND
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

PENETRATION OF THERMAL NEUTRONS IN CYLINDRICAL DUCTS THROUGH HYDROGENOUS MEDIA

A. Etemad and J.M. Pictet
Swiss Federal Institute for Reactor Research
Würenlingen / Switzerland

This paper presents an attempt to fill the gap between simple methods of thermal flux evaluation in straight ducts [1] giving generally poor results and prohibitively time consuming Monte Carlo calculations.

A method for the determination of the thermal neutron flux in a duct, due to neutrons of all energies emitted by a plane source perpendicular to the axis of the duct (Fig. 1), is elaborated.

The thermal flux in the duct is divided into three components (Fig. 1):

a) direct component Φ_a,

b) reflected component Φ_b,

c) leakage component Φ_c.

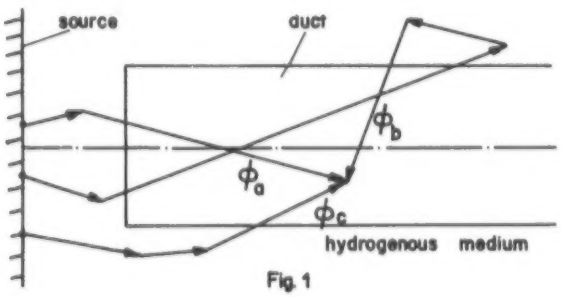

Fig. 1

The determination of the first component presents no difficulty. The main problem concerns the determination of the other components.

The method is based on the application of transport theory inside the duct and two-group diffusion theory in the medium surrounding the duct. The connection between these theories is effected by a proper choice of angular distributions for the neutrons entering the duct.

Translation provided by courtesy of the submitting Government

It is assumed that there is equilibrium between fast and thermal fluxes in the neighbourhood of the duct implying a sufficient hydrogen content in the medium. This assumption introduces a major simplification in the method by allowing the calculations to be made for the thermal group only.

The boundary condition at the duct's wall is satisfied by performing accurate book-keeping of neutrons at each point taking account of multiple reflection of neutrons by the medium surrounding the duct. This leads us to an integral equation of the Fredholm type, the solution of which constitutes the most important part of the numerical effort.

As an example the method is applied to the case of ducts through water.

In order to check the validity of this method, thermal neutron flux measurements by dysprosium foils are performed in tightly sealed plexiglas tubes immersed in the pool of the swimming pool reactor SAPHIR [2] using the core of the reactor as the neutron source.

The comparison between the calculated and the measured fluxes in two ducts of different dimensions shows that the theoretical results are in a very good agreement with the experimental results. The maximum relative deviation for a duct 250 cm in length by 12.5 cm in diameter is 14% and for a duct 170 cm in length by 5.9 cm in diameter is 22%. The maximum neutron flux attenuations in the ducts are respectively $6 \cdot 10^{-4}$ and $2 \cdot 10^{-4}$.

[1] Formulaire sur le calcul de la protection des réacteurs, Rapport C.E.A. No. 2253, 1963

[2] F. Alder et al., "Description of the Swiss Research Reactors", RAG-Bericht Nr. 2, 1957